BWR Vessel and Internals Project
BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74NP)

BWR Vessel and Internals Project BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74NP)

TR-113596NP

Final Report, February 2000

EPRI Project Manager
R. Thomas

EPRI • 3412 Hillview Avenue, Palo Alto, California 94304 • PO Box 10412, Palo Alto, California 94303 • USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com

DISCLAIMER OF WARRANTIES AND LIMITATION OF LIABILITIES

ORGANIZATION(S) THAT PREPARED THIS REPORT:

GE Nuclear Energy

ORDERING INFORMATION

Requests for copies of this report should be directed to the EPRI Distribution Center, 207 Coggins Drive, P.O. Box 23205, Pleasant Hill, CA 94523, (510) 934-4212.

CITATIONS

This report was prepared by

GE Nuclear Energy
175 Curtner Avenue
San Jose, CA 95125

Author
R. M. Horn

GE Principal Investigators
B. J. Branlund
D. O. Henry
R. M. Horn
H. S. Mehta
L. J. Tilly

Prepared for Boiling Water Reactor Vessel and Internals Project and EPRI.

The report is a corporate document that should be cited in the literature in the following manner:

BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74NP), EPRI, Palo Alto, CA, and BWRVIP: 2000. TR-113596.

REPORT SUMMARY

The Boiling Water Reactor Vessel and Internals Project (BWRVIP), formed in June, 1994, is an association of utilities focused exclusively on BWR vessel and internals issues. This BWRVIP report provides guidelines for inspecting and evaluating BWR reactor pressure vessels (RPVs).

Background
BWRs and their associated components are subject to degradation from a number of mechanisms ranging from general corrosion to fatigue cycling. To ensure that components retain their functionality, inservice inspections are performed on a regular basis. The requirements for these inspections are documented in a number of places, including the American Society of Mechanical Engineers (ASME) Code and BWRVIP Reports.

Objective
- To bring together inspection recommendations found in various documents.
- To demonstrate that these recommendations are adequate for managing all aging mechanisms for the current licensing term and for a 20-year license renewal term.

Approach
The project team performed a comprehensive review of all applicable documents as well as ASME Code and BWRVIP inspection requirements applicable to RPVs. The report contains a summary of these documents. To determine the adequacy of inspection recommendations in ensuring continued RPV integrity, the team reviewed recommendations with respect to known degradation mechanisms (as well as degradation history).

Results
The report concludes that existing inspection recommendations are, indeed, adequate to ensure RPV integrity for current plant design life as well as an additional 20-year period of life extension. Specific details regarding the license renewal period are discussed in an Appendix to the report.

EPRI Perspective
The BWRVIP has elected to present the report to the NRC in the form of a BWRVIP Inspection and Flaw Evaluation Guideline. By compiling the information in one report—with references to other BWRVIP and industry reports—the NRC can quickly evaluate the near term and license renewal requirements together. BWRVIP's intention is to promote the philosophy that the renewal term is a simple extension of the current term.

Keywords
Boiling water reactor
Flaw evaluation
Inspection strategy
Reactor pressure vessel
Stress corrosion cracking
Vessel and internals
Neutron embrittlement
License renewal

ACKNOWLEDGMENTS

The members of the BWRVIP Vessel I&E Guideline Focus Group, listed below, are gratefully acknowledged for their efforts which led to the successful completion of this document.

C. Garrow	BECO
C. Griffin	CP&L
K. Hacker	FTI
S. Leshnoff	GPUN
T. Pickens	NSP
C. Pierce	SNC
F. Polaski	PECO
B. Syx	SNC
R. Thomas	EPRI

General Electric Nuclear Energy Principal Investigators:

B. J. Branlund
D. O. Henry
R. M. Horn
H. S. Mehta
L. J. Tilly

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

EXECUTIVE SUMMARY

The purpose of this Reactor Pressure Vessel (RPV) Inspection and Evaluation Guideline report is to compile background understanding and to summarize inspection programs that are currently applicable to all of the RPV components. The inspection programs include (1) the American Society of Mechanical Engineering (ASME) Pressure Vessel Code established procedures and guidelines for the regular inspection of pressure boundary components in its Section XI program and (2) the Boiling Water Reactor Vessel Internals Project (BWRVIP) developed Inspection and Evaluation Guidelines that are specific to the pressure boundary internal components.

The report first reviews applicable documents that are currently applicable to the RPV components. The report then gives general descriptions of the different components and covers all of the potential degradation mechanisms that could be relevant to the various components. These sections give direct references to other reports that are sources of greater detail on specific plant configurations and mechanisms. The report then screens the mechanisms to establish which are relevant to BWR vessel components.

The next section of the report then reviews all of the existing ASME and BWRVIP inspection requirements that are currently being followed.

Content Deleted -
EPRI Proprietary Information

1 INTRODUCTION AND OBJECTIVE

1.1 Background

Boiling Water Reactors (BWRs) and their associated components are subject to various degradation processes ranging from general corrosion to fatigue cycling. In order to assure that the components retain their full functionality, inservice inspections are performed on a regular basis. The frequency of inspections in the initial 40-year operating period, as well as in a license renewal period, should be adequate to detect the different types of degradation mechanisms which, in turn, vary, depending on the material and operational environment. The American Society of Mechanical Engineering (ASME) Pressure Vessel Code has clearly established set procedures and guidelines for the regular inspection of pressure boundary components in its Section XI [1]. Complementing these requirements, the Boiling Water Reactor Vessel Internals Project (BWRVIP) has undertaken several extensive studies to establish guidelines for the inspection and evaluation of all safety-related reactor internal components and structures of the BWR type reactors. These studies were based on the results of an overall safety assessment report on the internals [2] which identified all the safety-related components and their configuration. The BWRVIP proceeded to develop specific Inspection and Evaluation Guidelines for all of these internal components [3-13]. Figure 1-1 schematically shows the locations of these components and highlights the appropriate document prepared. Table 1-1 lists the titles of the different BWRVIP documents shown in Figure 1-1. The comprehensive effort has addressed all of the key internal components, many of which are attached to the reactor pressure vessel (RPV) itself.

The NRC, nuclear industry and EPRI have performed extensive programs to address key vessel components such as the feedwater nozzles and the recirculation piping and nozzles [14-19]. These efforts provide significant guidance as to the inspection of these key structures, and document mitigation technologies to reduce the potential for degradation [20-22].

The industry has also undertaken studies to document the different degradation mechanisms that are relevant to the RPV as they apply to license renewal [23]. These studies are useful in aiding the industry to understand the inspection needs for any specific component. One of the most important time limiting aging mechanisms is neutron embrittlement of the RPV in the high fluence zones. The industry and the NRC have continued to update their understanding of the degradation processes and how they relate to the specific material (base and weld) by composition and process type. Again, these efforts are well documented [24-26]. In the area of fatigue, EPRI and the industry have also worked to better understand the cyclic loading events at actual BWRs [27, 28].

Content Deleted -
EPRI Proprietary Information

1.2 Objectives and Scope

The objective of the Reactor Pressure Vessel Inspection and Evaluation Guideline is to bring together all of this previous information in a consistent way with the intent of demonstrating that aging mechanisms are adequately managed for the current licensing term and for the 20-year license renewal term. Figure 1-2 presents a roadmap to the structure of the guideline including the process to review the needs for supplemental inspections or evaluations. The report will first review the general descriptions of the different components and will list all the potential degradation mechanisms that could be relevant to the RPV components. These sections will provide direct references to other reports as a source of greater detail on specific plant configurations. The report will screen the mechanisms to establish which are relevant to BWR vessel components. The report then reviews the existing ASME and BWRVIP inspection requirements.

Content Deleted -
EPRI Proprietary Information

The report includes an Appendix for License Renewal. The license renewal period builds on the current understanding and inspection programs. The extended period of operation will require the updating of all time limiting aging assessments (TLAAs) associated with the original plant design bases. These TLAAs will be divided into those that can be addressed using generic evaluations or those that require individual plant assessments. The report will provide a process to address these updates required for license renewal. The appendix also presents an update of the Equivalent Margin Analysis for the upper shelf energy that can be used generically for all the types of RPV plate materials and welds in the different BWR plants. Figure 1-2 also contains the license renewal related steps.

The BWRVIP has chosen to present the information in the form of a BWRVIP I&E Guideline to aid the NRC in its review. By compiling the information in one report with references to the earlier BWRVIP and industry reports, the NRC can quickly evaluate the near term and license renewal requirements together. It is the intention of the BWRVIP to promote the philosophy that the renewal term is a simple extension of the current term.

The RP Vessel I&E guideline will be applicable to all U.S. BWR/2-6 plants as well as the Spanish and Mexican BWRs. The components addressed will include the vessel shell, top and bottom heads, closure flanges and studs, support skirt, nozzles, safe ends, penetrations, internal and external attachments, in-core monitor housings, CRD stub tubes, and pressure boundary portions of CRD housings. The ASME Code 1989 edition will be used for guideline preparation in that it is referenced in the latest revision of 10CFR50.55a, the 1997 revision. In addition, the 1995 edition will be used as it pertains to inspection methods.

The work in this report was performed in accordance with 10CFR50, Appendix B.

Table 1-1
BWRVIP Documents Covering Inspection and Evaluation of Reactor Internals Components

Reactor Internal Component	BWRVIP I&E Number	BWRVIP Document Title
Core Shroud	BWRVIP-01	"BWR Core Shroud Inspection and Evaluation Guidelines, Revision 2," EPRI TR-107079, October 1996.
Core Spray	BWRVIP-18	"BWR Core Spray Internals Inspection and Evaluation Guidelines, " EPRI TR-106740, July 1996.
Shroud Support	BWRVIP-38	"BWR Shroud Support I&E Guidelines," EPRI Report TR-108823, September 1997.
Top Guide	BWRVIP-26	"BWR Top Guide Inspection and Evaluation Guidelines, " EPRI TR-107285, December 1996.
Core Plate	BWRVIP-25	"BWR Core Plate Inspection and Evaluation Guidelines, " EPRI TR-107284, December 1996.
Standby Liquid Control	BWRVIP-27	"BWR Standby Liquid Control System/Core Plate ΔP I&E Guidelines," EPRI Report TR-107286, April 1997.
Jet Pump Assembly	BWRVIP-41	"BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines," EPRI Report TR-108728, October 1997.
CRD Guide/Stub Tube	BWRVIP-47	"BWR Lower Plenum I&E Guidelines," EPRI Report TR-108727, December 1997.
In-core Housing/Dry Tube	BWRVIP-47	"BWR Lower Plenum I&E Guidelines," EPRI Report TR-108727, December 1997.
Instrument Penetrations	BWRVIP-49	"BWR Instrument Penetrations I&E Guidelines," EPRI Report TR-108695, March 1998.
LPCI Coupling	BWRVIP-42	"LPCI Coupling Inspection and Evaluation Guidelines, " EPRI TR-108726, September 1997.
RPV Circumferential	BWRVIP-05	"BWR RPV Shell Weld Inspection Recommendations," EPRI Report TR-105697, September 1995.
Vessel ID Brackets	BWRVIP-48	"BWRVIP, BWR Vessel ID Attachment Weld I&E Guidelines," EPRI Report TR-108724, February 1998.
Shroud Re-inspection	BWRVIP-07	"Guidelines for Reinspection of BWR Core Shrouds," EPRI Report TR-105707, February 1996.
Shroud Vertical Weld Inspection	BWRVIP-63	"Shroud Vertical Weld Inspection and Evaluation Guidelines," EPRI Report TR-113170, June 1999.

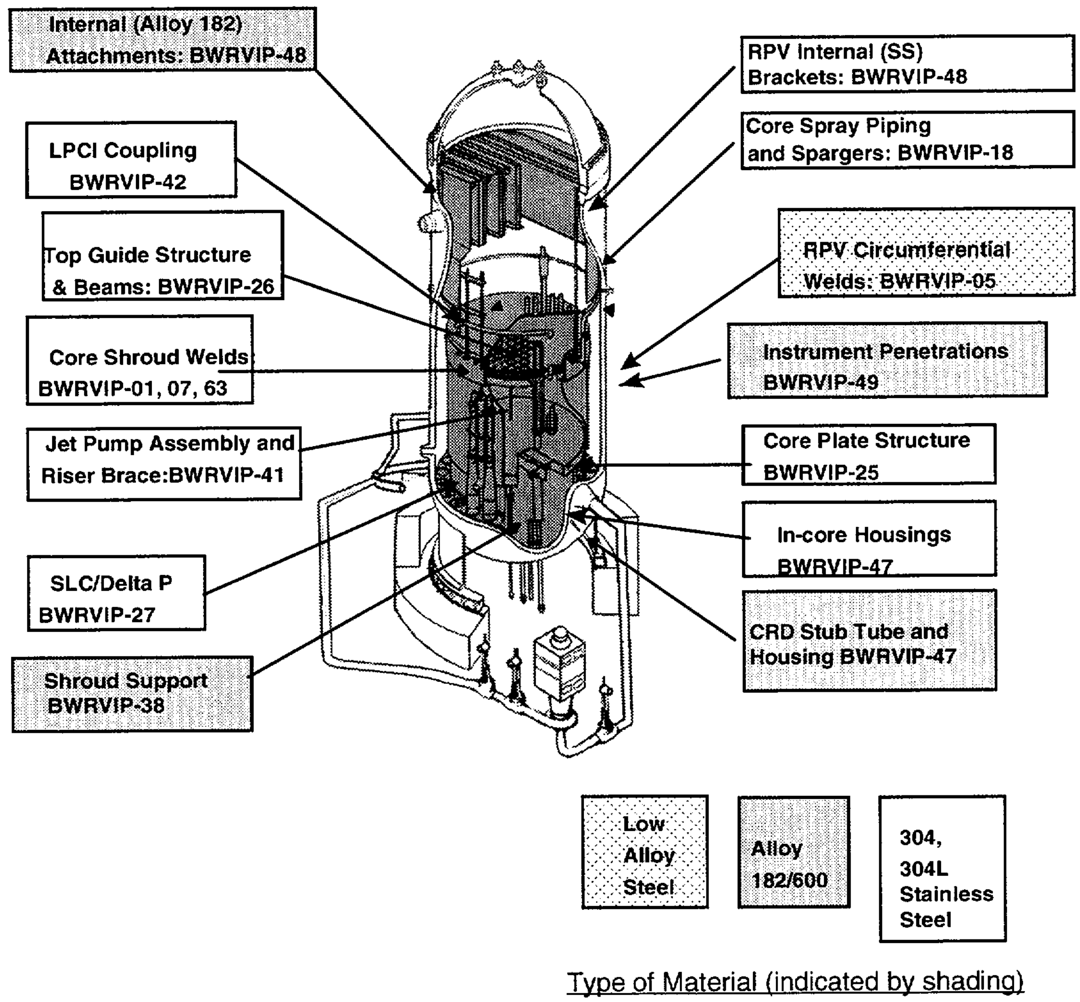

Figure 1-1
Schematic of RPV and Reactor Internals with Appropriate BWRVIP Guidelines

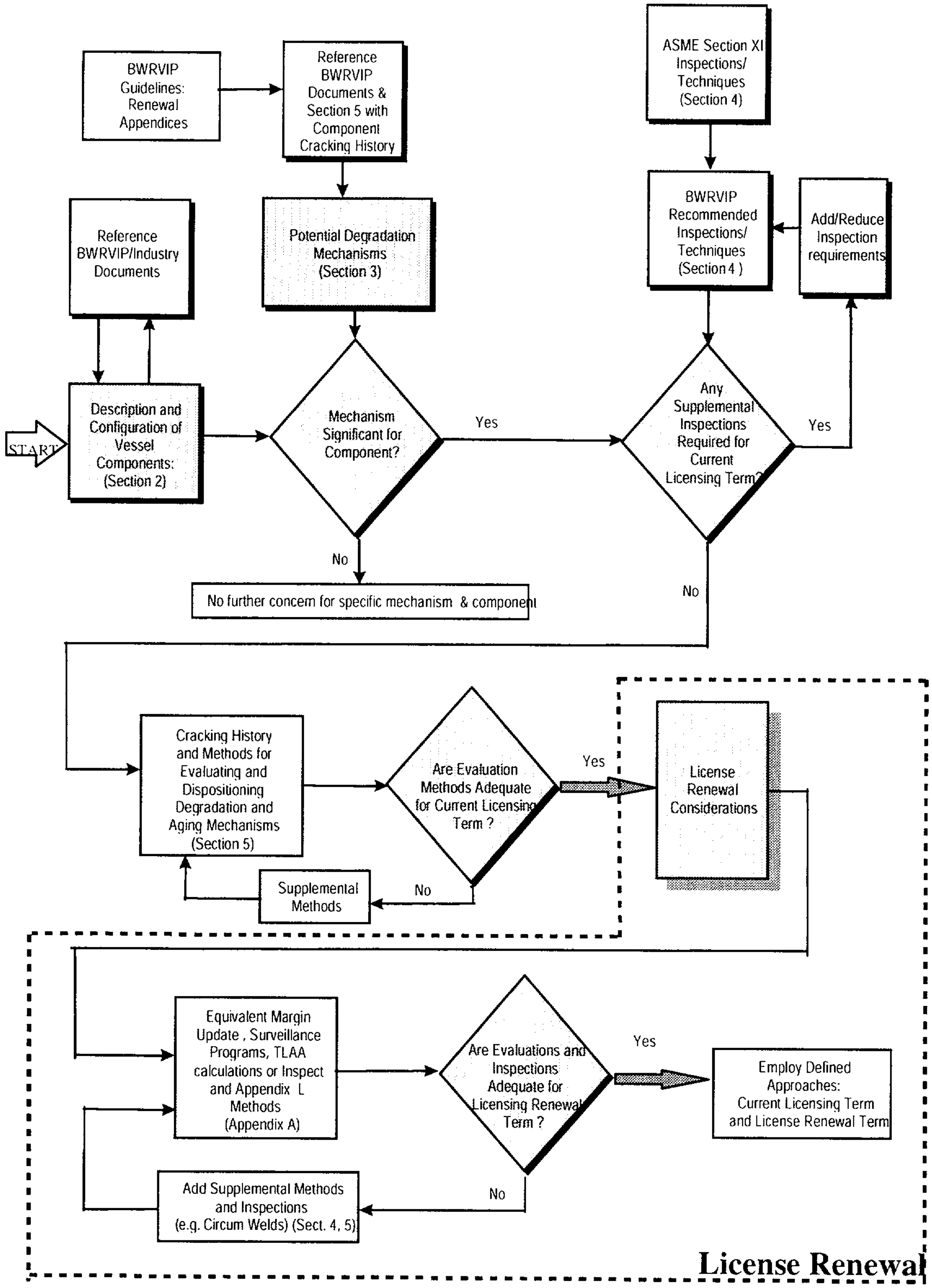

Figure 1-2
Vessel Inspection and Evaluation Guidelines Roadmap

2 GENERAL DESCRIPTION OF VESSEL COMPONENTS

The major RPV components are described in several documents [23, 29-30]. The RPV is comprised of a shell and a removable top head (flanged and joined via the head-to-flange closure studs), closure studs, a bottom head which is welded to the shell, multiple nozzles and safe-ends, multiple penetrations and control rod drive stub tubes, a vessel support skirt and numerous attachment welds. BWR vessels (within the scope of this report) have the following ranges of dimensions: the inside diameters (I.D.s) range from 183 to 251 inches, the vessel shell minimum thicknesses from 4.5 to 7.1 inches, the vessel heights (inside top head to inside bottom head) from 758 to 872 inches and the vessel head thicknesses from 2.7 to 6.8 inches. Reference 23 provides representative dimensions and identifies fabricators for the vessels of all U.S. BWR plants from BWR/2 through BWR/6. Figure 2-1 shows a typical RPV with all of the pressure boundary components.

General Electric Company nuclear steam supply product lines BWR/2 through BWR/6 represent an evolutionary development of RPV design concepts.

Content Deleted -
EPRI Proprietary Information

2.1 Reactor Pressure Vessel Shell, Top Head and Bottom Head

BWR vessels have been manufactured using different materials for the different components (shells, nozzles, flanges, studs, etc.). In addition, the choice of construction materials changed as the BWR product line evolved and newer materials were added to the ASME code.

Content Deleted -
EPRI Proprietary Information

Content Deleted -
EPRI Proprietary Information

2.2 Closure Flange and Studs

Content Deleted -
EPRI Proprietary Information

2.3 Vessel Support Skirt

Content Deleted -
EPRI Proprietary Information

2.4 Vessel Nozzles and Safe-ends

Reactor vessels have many penetrations for piping and equipment.

Content Deleted -
EPRI Proprietary Information

CRD penetrations and flux monitor penetrations have different designs on different vessels and are discussed in a following section. Figure 2-4 through 2-6 show typical configurations for the recirculation outlet, core spray and feedwater nozzles along with the safe ends, respectively.

Safe ends are attached to each of the nozzles providing a transition to the piping material. The material of the safe-end varies from system to system as well from plant to plant which is a result of the RPV vendor's typical practice.

Content Deleted -
EPRI Proprietary Information

**Content Deleted -
EPRI Proprietary Information**

2.5 Vessel Internal Attachments

The RPV attachments include the brackets used to hold internal components such as the core spray piping and the jet pump riser piping as well as the shroud support structure. These attachments have been fully characterized in BWRVIP-38 and BWRVIP-48.

**Content Deleted -
EPRI Proprietary Information**

BWRVIP-48 details the different configurations of both types of attachments. Figure 2-9 shows the details of an example core spray attachment bracket.

**Content Deleted -
EPRI Proprietary Information**

2.6 Vessel Instrument Penetrations

Several different penetrations must be considered, including the instrument nozzles required to measure water level (which are covered in BWRVIP-49) and the core plate ΔP/standby liquid control SLC) system (BWRVIP-27), which can also be a full penetration nozzle design, similar to those discussed in Section 2.4. The typical geometry is shown in Figure 2-11.

**Content Deleted -
EPRI Proprietary Information**

2.7 Vessel In-core Housing, CRD Stub Tube and CRD Housing Penetrations

The configurations of the in-core monitor housings (ICMH), CRD stub tubes and CRD housings are given in BWRVIP-47. (Tables 2.1-1, -2, -5, -6, Figures 2.1.1-1 through 2.1.1-7 and Figures 2.1.2-1 through 2.1.2-4 of BWRVIP-47 give the details for the specific plants.) Typical geometries are displayed in Figures 2-13 and 2-14 of this report.

**Content Deleted -
EPRI Proprietary Information**

Content Deleted -
EPRI Proprietary Information

2.8 Vessel External Attachments

In addition to the support skirt and the closure studs, there are other attachments to the outside of the vessel, including the Top Head Lifting Lugs, Stabilizer Brackets, Insulation Brackets and Refueling Bellows. All of these attachments are exposed only to an air environment.

Content Deleted -
EPRI Proprietary Information

Table 2-1
Typical Materials Used for RPV Components (Reference 23)

Content Deleted -
EPRI Proprietary Information

Table 2-2
Typical Number and Types of Nozzles for the Different BWR Designs (Reference 23)

Content Deleted -
EPRI Proprietary Information

Table 2-3
Typical Nozzle Safe End Materials Summary (Reference 23)

**Content Deleted -
EPRI Proprietary Information**

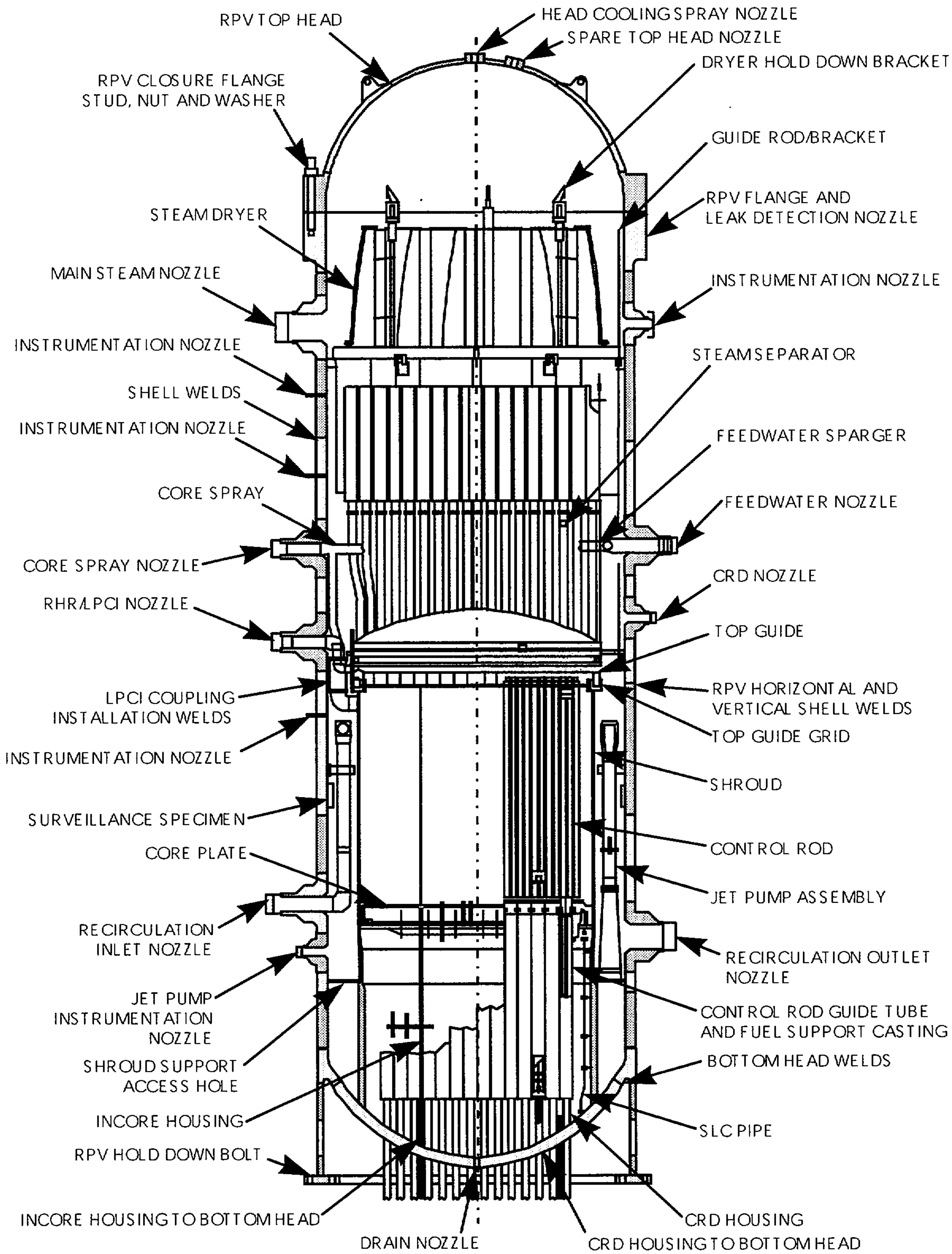

Figure 2-1
Typical Reactor Pressure Vessel - Nozzles and Attachments Shown Schematically

Content Deleted -
EPRI Proprietary Information

Figure 2-2
Schematic of Typical configurations for (a) the RPV Top Head with Flange and (b) the Closure Stud

Content Deleted -
EPRI Proprietary Information

Figure 2-3
Detail of Typical RPV Support Skirt

Content Deleted -
EPRI Proprietary Information

Figure 2-4
Typical Recirculation Outlet Nozzle, Nozzle Butter, Weld and Safe End

Content Deleted -
EPRI Proprietary Information

Figure 2-5
Typical Core Spray Nozzle, Nozzle Butter, Weld and Safe End

Content Deleted -
EPRI Proprietary Information

Figure 2-6
Typical Feedwater Nozzle, Nozzle Butter, Weld and Safe End

Content Deleted -
EPRI Proprietary Information

Figure 2-7
Typical Shroud Support Structure with Leg Design
(The H9 and H12 welds which join the component to the RPV are also shown.)

**Content Deleted -
EPRI Proprietary Information**

**Figure 2-8
Typical Shroud Support Structure with Gussets**

**Content Deleted -
EPRI Proprietary Information**

Figure 2-9
Typical Core Spray Bracket Attachment Weld

Content Deleted -
EPRI Proprietary Information

Figure 2-10
Typical Jet Pump Riser Attachment Weld

Content Deleted -
EPRI Proprietary Information

Figure 2-11
Typical Instrument Penetration Component with Weld

Content Deleted -
EPRI Proprietary Information

Figure 2-12
Typical Core DP/SLC Nozzle and Welds

Content Deleted -
EPRI Proprietary Information

Figure 2-13
Typical In-Core Monitor Housing with RPV Weld and Housing to Guide Tube Weld

Content Deleted -
EPRI Proprietary Information

Figure 2-14
Typical CRD Stub Tube and CRD Housing Configurations

3
POTENTIAL DEGRADATION MECHANISMS

The degradation mechanisms that are considered in this report are neutron irradiation embrittlement, low-cycle and high-cycle fatigue, including both mechanical and thermal effects, intergranular stress corrosion cracking (IGSCC), irradiation-assisted stress corrosion cracking (IASCC), stress corrosion cracking (SCC), general corrosion, flow-assisted corrosion (erosion-corrosion) and thermal embrittlement. These mechanisms have been identified from a review/evaluation of nuclear power plant operating experience, relevant laboratory data, and related experience in other industries. They are also discussed in earlier industry reports [23], NRC documents [14,15] and BWRVIP references [3-13, 16]. Each will be presented in the following sections with references to other comprehensive discussions in the related references.

3.1 Neutron Embrittlement

3.1.1 Mechanism Review

Neutrons produce energetic primary recoil atoms which displace large numbers of atoms from their crystal lattice positions in metallic materials by a chain of atomic collisions. Neutron exposure damage can be characterized by displacements per atom (dpa), which accounts for the neutron energy spectrum as well as the fluence. These displacements produce defects that change the properties of the metal.

Content Deleted -
EPRI Proprietary Information

3.1.2 Locations of Significant Importance

**Content Deleted -
EPRI Proprietary Information**

3.2 Fatigue

3.2.1 Mechanism Review

Fatigue is the initiation of cracks and subcritical crack growth under the influence of fluctuating or cyclic applied stresses. Various sources exist for fluctuating stresses, but the chief sources are vibration and temperature fluctuations. Fatigue behavior of components is affected by a variety of parameters such as stress range, mean stress frequency, surface roughness, and environmental conditions.

**Content Deleted -
EPRI Proprietary Information**

The specific case of environmental fatigue has been considered. The necessary information has been developed by the industry to determine the effects of BWR water environmental conditions on fatigue life. T

**Content Deleted -
EPRI Proprietary Information**

One important source of cyclic stresses can be the mixing of hot and cold water. The potential for thermal fatigue depends upon the temperature difference between the mixing fluids, flow velocities and heat transfer coefficients.

**Content Deleted -
EPRI Proprietary Information**

Content Deleted -
EPRI Proprietary Information

3.2.2 Locations of Significance

3.2.2.1 Closure studs

Content Deleted -
EPRI Proprietary Information

3.2.2.2 Penetrations

Content Deleted -
EPRI Proprietary Information

3.2.2.3 Safe-ends

Content Deleted -
EPRI Proprietary Information

3.2.2.4 Nozzles

Content Deleted -
EPRI Proprietary Information

Content Deleted -
EPRI Proprietary Information

3.2.2.5 Vessel Support Skirt

Content Deleted -
EPRI Proprietary Information

3.2.2.6 Vessel External Attachments

Content Deleted -
EPRI Proprietary Information

3.3 Intergranular Stress Corrosion Cracking (IGSCC)

Stress corrosion cracking (SCC) is the degradation category associated with the initiation and growth of cracks in a susceptible alloy system under the influence of three factors: tensile stress, susceptible material, and a corrosive environment.

Tensile stresses causing SCC are typically at or above the material's yield strength levels. It is very much a function of the material condition and the environment. The first category of SCC which relates to austenitic structural materials such as wrought stainless steels or nickel base materials is intergranular stress corrosion cracking (IGSCC). Technically, cracking occurs along grain boundaries in wrought alloys or along dendrite boundaries in weld metal (the governing microstructural feature in welds). Cracking is promoted by the BWR normal water chemistry (NWC) environment, which contains highly oxidizing species such as oxygen and hydrogen peroxide.

Content Deleted -
EPRI Proprietary Information

**Content Deleted -
EPRI Proprietary Information**

3.3.1 Locations of Significance

3.3.1.1 Nozzle-to-safe end welds

**Content Deleted -
EPRI Proprietary Information**

3.3.1.2 Attachment Welds

**Content Deleted -
EPRI Proprietary Information**

3.3.1.3 Instrument Penetrations, In-core Housings and CRD Stub Tubes/Housings

**Content Deleted -
EPRI Proprietary Information**

3.3.1.4 Closure Studs

Content Deleted -
EPRI Proprietary Information

3.4 Irradiation-Assisted Stress Corrosion Cracking (IASCC)

Irradiation-assisted stress corrosion cracking (IASCC) is very closely related to IGSCC [39]. The major differences are the influences of the irradiation (neutron flux) on both the material's susceptibility and the residual stresses. The flux is already important in changing the core environment by producing the oxidizing species through radiolysis. The neutrons interact with the material enhancing the diffusion processes in the metal and producing matrix hardening.

Content Deleted -
EPRI Proprietary Information

3.5 Stress Corrosion Cracking (SCC) of Low Alloy Steel

The potential for SCC of low alloy steel in BWRs has been addressed over 20 years. These materials have exhibited excellent performance as documented in the BWRVIP-60 report on low alloy steel crack growth rates [40]. Only under very specialized conditions in the laboratory can the sustained SCC growth rates in these alloys be high.

Content Deleted -
EPRI Proprietary Information

**Content Deleted -
EPRI Proprietary Information**

3.6 General Corrosion

Corrosion, discussed in earlier reports [23], is the electrochemical reaction between a metal or alloy and its environment. It is characterized by the loss of material or the deterioration of surfaces such that the material's properties or load bearing capability are changed. These effects can be categorized in two ways.

**Content Deleted -
EPRI Proprietary Information**

3.7 Flow-Accelerated Corrosion (Erosion/Corrosion)

Erosion/corrosion, also commonly called flow accelerated corrosion (FAC), is a special form of corrosion caused by flowing liquids. The flow can lead to cavitation, which, in turn, can disrupt surface oxide films and accelerate corrosion of the base metal, leading to a loss of material.

**Content Deleted -
EPRI Proprietary Information**

3.8 Thermal Embrittlement

There is the potential for the degradation of the fracture toughness of cast austenitic stainless steel through thermal aging processes. The presence of significant amounts of ferritic phase with its higher Cr content could lead to a reduction in fracture toughness in the ferritic phase due to sigma phase precipitation.

**Content Deleted -
EPRI Proprietary Information**

3.9 Summary of Applicability of Degradation Mechanisms

Based on the discussion presented, the significant degradation mechanisms are neutron embrittlement, fatigue and IGSCC. This is summarized in Table 3-1 by component and mechanism. The inspection requirements and surveillance programs for both the present licensing period as well as any license renewal period should be designed to address detection of the effects of these mechanisms, either the loss of toughness or the presence of cracking, in these components at the relevant locations.

Table 3-1
BWR RPV Aging Mechanism Assessment Summary

Content Deleted -
EPRI Proprietary Information

Table 3-1
BWR RPV Aging Mechanism Assessment Summary (concluded)

Content Deleted -
EPRI Proprietary Information

4 REVIEW OF INSPECTION REQUIREMENTS

4.1 ASME Section XI Preservice Inspections

The ASME Code Section XI "Rules for Inservice Inspection of Nuclear Power Plant Components" was first published in 1970 and specified requirements for Preservice examinations as well as in-service examinations. The ASME Section XI Preservice examinations differed from the examinations required by the Construction Code because they were intended not only to verify the vessel integrity, but also to form a baseline for subsequent inservice examinations. Preservice examinations of plants constructed before that time included only those of the Construction Code, which would have been Section III of the ASME Code after its first publication in 1964. Prior to 1964, Section VIII of the ASME Code was used for construction with Code Cases which described additional quality assurance requirements for nuclear vessels. This report reviews the preservice and inservice examination requirements for the reactor vessel. BWRVIP-05 [16] provides a more detailed treatment of vessel ultrasonic examination methodology and describes the evolution of the preservice examination requirements of ASME Section XI from the 1970 Edition to the 1989 Edition. As a point of reference, the terminology of the 1989 Edition of Section XI of ASME Code has been used in preparation of this report, since it was the Edition of record cited by the NRC in Title 10 CFR 50 Section 50.55a, at the time this report was prepared.

The ASME Code, Section XI requires that all mandated examinations, except for the VT-2 examinations (visual examinations for leakage which are also required), be performed once prior to initial plant startup and then periodically after plant startup as inservice examinations. The required examinations are listed in ASME Section XI, Table IWB-2500-1 for the applicable examination categories. For the reactor vessel, the applicable categories are:

B-A	Reactor Vessel Seam Welds
B-D	Reactor Vessel Full Penetration Nozzle Welds
B-E	Partial Penetration Nozzle Welds (CRD, ICM, SLC)
B-F	Dissimilar Metal Welds (Safe-Ends)
B-J	Piping Welds (Safe Ends for Feedwater and Main Steam)
B-G-1	Bolting Greater than 2 in. in Diameter (Closure Head Studs)
B-G-2	Bolting less than 2 in. in Diameter (Top Head Flange/CRD housing bolts)
B-H	Exterior Attachment Welds
B-N-1	Interior of Reactor Vessel
B-N-2	Core Support Structure and Internal Attachment Welds
B-P	Pressure-Retaining Components

4.1.1 Examination Methods

In addition to identifying the items to be examined, the ASME Table IWB-2500-1 identifies the examination method to be used for the examination. ASME Section XI breaks examination requirements into three categories: visual, surface and volumetric. Visual examinations consist of three categories: (1) VT-1, a detailed visual examination including a lighting resolution check; (2) VT-2, an examination for leakage; and (3) VT-3, an examination for general structural and mechanical integrity. Additional resolution requirements for visual examination were added to Section XI in the 1990 Addenda, increasing the VT-1 resolution requirement and, for the first time, imposing resolution requirements for VT-2 and VT-3. Surface examination may be either magnetic particle or liquid penetrant examination. Volumetric examination technically includes both radiography and ultrasonic examination; however, for practical reasons, ultrasonic examination is preferred for inservice examination, so it is also used to perform the preservice examinations.

ASME Section XI IWB-2200 allows shop and field examinations performed during fabrication and construction to be accepted as a preservice examination, provided (1) the examinations are performed using the same technique as would be used in subsequent inservice examinations and (2) the examinations are performed after the Construction Code (ASME Section III) hydrostatic test. Although such credit was taken for some shop and field examinations, the credit was usually limited to the surface examinations, since radiography is used almost exclusively in ASME Section III for the volumetric examination of full penetration welds. The preservice examinations were performed using an ultrasonic method which was consistent with the practice that would be used for subsequent inservice examinations. The result was that prior to operation, the vessel weld seam and full penetration nozzle welds were radiographed under Construction Code rules (Section III) and then examined again ultrasonically to Section XI rules. Note that some early vessels (BWR-1, BWR-2) and some BWR-3 vessels, were placed in operation before industry implementation of Section XI and were not examined ultrasonically before service, but have been examined ultrasonically inservice under Section XI rules. In 1983, US NRC Regulatory Guide 1.150, Revision 1 [44] was issued which included a number of recommendations, including increased recording sensitivity of the inservice ultrasonic examinations performed on reactor vessel welds beyond the requirements existing at that time in ASME Section XI (and referenced requirements of Section V). This required the evaluation of more indications, including those which were determined to be minute anomalies such as fine segregates in the vessel plate material. Following the issuance of the Regulatory Guide, the increased recording sensitivity was applied to the preservice examination of vessel welds.

Since implementation of Regulatory Guide 1.150, improvements in examination methodology have been made to the Code. In the 1986 Addenda to Section XI, the recording sensitivity was increased to the same level as required by the Regulatory Guide. Then in the 1989 Addenda to ASME Section XI, quantitative performance demonstration requirements were added for ultrasonic examinations in Appendix VIII. Under the rules of Appendix VIII, procedures, equipment, and personnel must be demonstrated to achieve flaw detection and sizing results which exceed the requirements of earlier Editions of the Code. The development of these rules began in the early 1980s in response to NRC requirements imposed by Bulletins 82-03 [45] and

83-02 [46], which required demonstration of ultrasonic examination capability for welds in large bore reactor recirculation piping. However; the new rules in Appendix VIII extend the demonstrations to cover virtually all the inservice ultrasonic examinations required by the Code. While those requirements have not yet been fully mandated by the NRC, the utilities have formed an alliance, the Performance Demonstration Initiative (PDI), and, in conjunction with EPRI, have developed a program to implement those requirements. Many utilities are already using personnel, equipment, and procedures which have been qualified under the new performance demonstration program.

4.1.2 Examination Requirements for Specific Categories

This section describes the examination requirements of ASME Section XI for the individual items identified by examination category. Since the preservice examinations are conducted using the same techniques as in later inservice examinations, the requirements described here apply to both preservice and inservice examinations. The following discussion applies to the preservice and inservice inspection program requirements as they appear in the 1989 Edition of ASME Section XI except where noted.

Reactor vessel seam welds (Category B-A) and Full Penetration Nozzle Welds (Category B-D, which includes also the nozzle inner radius region) are ultrasonically examined. Examinations of closure and bottom head welds are limited to "accessible lengths" under Code rules, acknowledging the inherent obstructions caused by nozzle penetrations, particularly in the bottom head region. Safe-end welds (Category B-F and B-J) receive both ultrasonic and a surface examination if they connect to lines over 4 inches in nominal pipe size, which include main steam, feedwater, core spray and all of the reactor recirculation nozzles. Safe-end welds connecting to lines smaller than 4 inches in nominal pipe size require surface examination. Partial penetration welds (Category B-E), such as used for control rod drive (CRD) and instrument penetrations, require only visual (VT-2) examination and therefore are not subject to a preservice examination. However, these weld joints require progressive surface examination during fabrication under ASME Section III to assure integrity through the weld volume. (Note that in IWB-2500-1 of ASME Section XI, the separate Category of B-E for partial penetration welded nozzles was deleted in the 1994 Addenda. since the same examination requirement (visual VT-2) is implemented in the same area under examination Category B-P, which applies to the entire reactor coolant pressure boundary.)

The bolting examination requirements are based on the bolting diameter. Vessel closure head studs (Category B-G-1, greater than 2 inches in diameter) require ultrasonic examination inservice when the examination is done in place, or both surface and ultrasonic examination if they are removed for examination. Smaller diameter bolting (B-G-2) on the vessel top head flange and CRD housing flanges on the bottom head are subject to visual, VT-1 examination.

Reactor vessel exterior attachments (Category B-H) require a surface examination (Note that in IWB-2500-1 of ASME Section XI, the separate category of B-H for vessel exterior attachments was deleted in the 1998 Addenda and the requirements for examination of vessel exterior attachments were included under category B-K along with those of piping, pumps, and valves.)

Reactor vessel interior attachments, as well the core support structure (both Category B-N-2), are subject to visual (VT-3) examination. Exceptions are made for attachments to the vessel wall in the beltline region, jet pump riser braces and surveillance sample holder brackets, which are subject to visual (VT-1) examination. In practice, the internal attachments for the shroud support, core spray internal piping, and jet pump riser braces are subject to a more rigorous visual examination requirement as described in other BWRVIP reports discussed later in this section. The reactor vessel interior itself (Category B-N-1) is also subject to a visual (VT-3) examination.

4.2 ASME Section XI: Table IWB-2500-1 Visual, Surface, and Volumetric Examinations; Categories B-A, B-D, B-E, B-F, B-G-1, B-G-2, B-H, B-N-1, B-N-2, B-P

Along with the required examinations methods, the frequency of inservice examinations for the various examination categories are also listed in ASME Section XI, Table IWB-2500-1. These categories were described in Section 4-1. These examinations are required to be performed once per interval, which is defined as 10 years. The interval may be extended up to one year to coincide with plant outages, but such extensions are not permitted to exceed more than one year during the plant lifetime (i.e., the extensions cannot be applied in such a way as to extend all the intervals to 11 years rather than 10 years). For some items, deferral of the required examinations, such as those for reactor seam welds, is permitted until the end of the interval. For most other examinations, such as those for piping (including safe-end welds) and bolting, some percentage of the examinations must be performed throughout the interval with requirements for each period within the 10-year interval as follows:

Period	Years (within each 10-year interval)	Minimum Exam % Completed	Maximum Exam % Credited
1st	0-3	16	34
2nd	3-7	50	67
3rd	7-10	100	100

For full penetration nozzle welds, 25%, but not more than 50%, of the examinations can be performed by the end of the first period and the remainder deferred until the end of the interval.

The IWB-2500-1 Table was revised in the 1989 Addenda to address plant life extension and specifies that the frequency of examination remains consistent for each successive 10-year inspection interval, regardless of the actual total number of years of plant operation.

4.3 NUREG-0619 - Feedwater Nozzle Examinations

US NRC NUREG-0619 "BWR Feedwater Nozzle and Control Rod Drive Return Drain Line Nozzle Cracking" [14] implemented "augmented" inservice examination requirements (i.e., examinations beyond those of ASME Section XI) on feedwater nozzle inner radius areas based on thermal fatigue cracking concerns. (The CRD return line nozzles in nearly all domestic plants which had them were capped, thus eliminating the concern for those components.) The NUREG provides recommendations for an increased frequency of ultrasonic examination (beyond ASME Section XI requirements) and recommended performing liquid penetrant examinations periodically (from every second to every ninth refueling outage, depending on the sparger design [14]). The nozzle inner radius examination area was also extended by the NUREG document beyond ASME Section XI requirements from the nozzle inner radius itself further into the nozzle bore. Performance of the liquid penetrant examination required drain-down of the vessel, decontamination, removal of the sparger and actual personnel contact with the nozzle inner radius from inside the vessel, resulting in high radiation exposure levels to workers. Subsequent improvements in ultrasonic examination techniques were made which were described in a BWR Owner's Group report submitted to the NRC [47]. Using the improved, automated UT techniques, detection of 0.25 in. deep flaws in the nozzle inner radius region was successfully demonstrated on nozzle mock-ups. Taking credit for the improvement, the report provided a "UT Inspection Interval Factor" based on that detection threshold to be used in conjunction with a plant-specific fracture mechanics postulated flaw growth analysis to develop a feedwater nozzle inner radius examination schedule appropriate for each plant. Based on these results, the NRC issued a Safety Evaluation Report in 1998 [48], allowing the improved, automated ultrasonic examinations to stand in lieu of the liquid penetrant examination and the frequency of the ultrasonic examinations to be reduced from the original NUREG requirements.

4.4 NUREG-0313 - Pressure Boundary Piping (As Applied to Safe-End Welds)

Generic Letter (GL) 88-01 [36], in conjunction with the basis provided in NUREG-0313, Revision 2, [15], implemented "augmented" inservice examination requirements (i.e., examinations beyond those of ASME Section XI) for stainless steel piping welds based on IGSCC concerns. The augmented program increased frequency of examination overall for 4" and larger diameter piping welds carrying primary system (reactor) water at temperatures above 200°F. With respect to the reactor vessel, these increased examination frequency requirements primarily affect safe-end-to-pipe welds where the original Grade 304 piping was not replaced with resistant low carbon grade material, such as 316L. NUREG 0313 also contained requirements for ultrasonic examination performance which apply to the examination of all stainless steel safe-end-to-nozzle welds (Category B-F). The NUREG requires that ultrasonic examination of the subject austenitic stainless steel welds be carried out using procedures and personnel qualified under the NDE Coordination plan agreed upon by NRC, EPRI, and the BWROG [49]. That plan originated in response to IE Bulletins 82-03 and 83-02 [45, 46], which were the predecessors of NUREG 0313, Revision 2 [14] and GL 88-01 [36]. This implemented an ultrasonic examination demonstration program, utilizing actual samples of pipe weldments (removed from operating plants) containing intergranular cracking to verify the adequacy of the technique and the skill of the examiner. The agreement has been subsequently modified to include the ASME Section XI, Appendix VIII implementation program currently being conducted by the PDI and EPRI.

4.5 BWRVIP-05 Requirements for Category B-A Seam Welds

The BWRVIP, in its report BWRVIP-05 [17], evaluated the current inspection requirements for the reactor pressure vessel shell welds in BWRs. It then formulated recommendations for alternative inspection requirements, with adequate technically-justified bases for these recommended requirements.

Content Deleted -
EPRI Proprietary Information

4.6 BWRVIP Reports Nos. 27, 38, 41, 47, 48, and 49

The BWR Vessel and Internals Project subcommittee of EPRI has published a number of reports recommending improvements in examination of reactor vessel internal components as discussed in Sections 1 and 2 [3-13]. This effort began in response to NRC Generic Letter 94-03 [51] requiring that consideration be given to improved examination technology following the discovery of cracking in reactor vessel core shroud welds. These recommendations included implementing visual examinations of higher resolution than those required by ASME Section XI. For purposes of this report, the BWRVIP reports focused on reactor internals with the only overlapping components of concern being the examination of attachments to the inside of the reactor vessel wall, those addressed by the ASME Code examination category B-N-2. The definitions and requirements for the higher resolution visual examinations were first published in BWRVIP-03 [52].

Content Deleted -
EPRI Proprietary Information

Content Deleted -
EPRI Proprietary Information

4.7 Summary Table of ASME/BWRVIP/NRC Recommended Inspections

Previous sections clearly establish that there are well defined and comprehensive inspection requirements for all of the vessel components. Table 4-1 provides a summary and cross-reference of these examinations required by the ASME Code and augmented examinations imposed by the NRC or as recommended in the various BWRVIP reports.

4.8 Adequacy of Integrated Inspection Requirements

As discussed in Section 3, there are four significant degradation mechanisms that are applicable to the vessel components: neutron embrittlement, fatigue, IGSCC and SCC of the closure studs. The inspection programs are directed at evaluating the vessel component surfaces for indications of crack initiation.

Content Deleted -
EPRI Proprietary Information

Table 4-1
Summary Table of ASME/BWRVIP/NRC Recommended Inspections

Content Deleted -
EPRI Proprietary Information

Table 4-1
Summary Table of ASME/BWRVIP/NRC Recommended Inspections (concluded)

Content Deleted -
EPRI Proprietary Information

5 REVIEW OF EVALUATION GUIDELINES

This section is a review of the different evaluation processes required for application with Vessel components. The first processes are the ASME code procedures for flaw evaluation and repair implementation. The other procedures are those that must be performed in accordance with 10CFR50 Appendix G.

The procedures for flaw evaluation and repair implementation are presented in Sections 5.1 and 5.2, respectively. A brief review of past vessel component inspection findings is presented in Section 5.3 and examples of the associated evaluations and repairs are presented in Sections 5.4 and 5.5, respectively. These examples complement those that have previously been given in other BWRVIP I&E Guidelines [8] and BWRVIP Crack Growth Rate Methodology Reports [40, 53].

The other procedures are those that must be performed in accordance with 10CFR50 Appendix G. These evaluations include the determination of the Upper Shelf Energy (USE) for the each of the specific plant RPV beltline materials. Also included is an evaluation of fracture toughness (i.e., an assessment of the RT_{NDT} adjusted for irradiation and the resultant P-T curves for plant operation). The USE evaluations are discussed in Section 5.6 and the fracture toughness evaluations are discussed in Section 5.7 along with an overview of surveillance programs.

5.1 Overview of ASME Section XI Flaw Evaluation Methodology

The ASME flaw evaluation procedures for nuclear pressure vessels are contained in ASME Section XI, Appendix A [54]; the original technical basis is documented in Reference 55 and an updated technical basis for corrosion fatigue crack growth is contained in Reference 56. ASME Section XI specifies periodic volumetric examinations of primary system pressure-retaining components that are generally accomplished using ultrasonic (UT) examination techniques, as discussed in Section 4.0. To simplify the analysis of indications detected by such examinations, Section XI adopted the conservative assumption that all observed indications are planar (crack-like) defects. The Code treats irregularly shaped flaws as idealized simple geometric shapes. UT inspection results are thus reduced to two characteristic flaw size parameters, a flaw depth (a or 2a) and flaw length (*l*), together with the relative location of the flaw in the component. Proximity limits are given for cases of adjacent multiple flaws and subsurface flaws that are close to the surface. Adjacent multiple flaws within the proximity limits are enveloped into a single longer flaw with new length and depth parameters. Subsurface flaws within the surface proximity limit are treated as surface flaws.

The flaw length and depth, as defined above, are compared directly to a set of acceptable standards defined in terms of the same geometric flaw parameters. Such standards reflect generic fracture analyses that define acceptable flaw sizes, below which no further evaluation is required. For flaw indications that exceed these standards, further detailed fracture mechanics evaluations are permitted.

Appendix A of Section XI provides a non-mandatory procedure for evaluating flaws in pressure vessels which exceed the acceptance standards. It consists of an application of linear elastic fracture mechanics (LEFM) to a given flaw using such factors as flaw shape, flaw orientation, flaw location, projected flaw growth and material fracture toughness.

The stress fields at the flaw location are determined from the vessel design stress report for normal (Level A), upset (Level B), emergency (Level C), and faulted (Level D) operating conditions; the normal operating condition included system testing. Crack tip stress intensity factors are determined for each operating condition and are used to project anticipated crack growth due to continued operation of the vessel. The stress intensity factor is also used to determine the critical size of the observed flaw at which it would exceed a conservative estimate of the material fracture toughness. For inside surface flaws, the flaw growth analysis must consider corrosion-fatigue due to the reactor operating environment, as shown in Appendix A of Section XI, Figure A-4300-1.

5.2 Overview of Section XI Repair Criteria

If the volumetric or surface examination reveals a flaw that exceeds the acceptance standards and cannot be shown to be acceptable by analytical evaluation, the flaw shall be either removed by mechanical methods or repaired to the extent necessary to meet the acceptance standards of IWB-3000. The repair and re-examination requirements are specified in IWA-4000.

5.3 Review and Summary of Past Vessel Component Inspection Findings

The past inspection findings in commercial BWR RPVs fall into five broad categories: (1) feedwater and CRD return line nozzle cracking; (2) nozzle safe-end cracking (Alloy 182 butter initiated); (3) RPV head cracking (head clad with austenitic stainless steel); (4) CRD stub tube cracking; and (5) attachment bracket cracking that could potentially grow into the RPV base metal. The details of these cracking incidents have been previously described in References 33 and 57 through 63. The recent BWRVIP reports also provide information on these findings [40, 53].

5.3.1 Feedwater and CRD Return Line Nozzle Cracking

In the late 1970s, inservice inspections (ISI) discovered cracking in BWRs on the inside surface of feedwater (FW) and control rod drive return line (CRDRL) nozzles.

Content Deleted -
EPRI Proprietary Information

Content Deleted -
EPRI Proprietary Information

Since the issuance of NUREG-0619, significant advances have occurred in UT inspection technology, and significant field experience has been gathered on the successful prevention of cracks in FW and CRDRL nozzles. As a result of these improvements, BWROG proposed that UT inspections replace the PT inspections in NUREG-0619, and that UT inspection intervals be based on sparger-sleeve configuration and specific UT inspection methods [45]. The NRC staff's acceptance of these recommendations is documented in Reference 46.

5.3.2 Nozzle Safe-End Cracking

Content Deleted -
EPRI Proprietary Information

Content Deleted -
EPRI Proprietary Information

5.3.3 RPV Head Cracking

Content Deleted -
EPRI Proprietary Information

Content Deleted -
EPRI Proprietary Information

5.3.4 CRD Stub Tube Cracking

Stub tubes provide a means of connecting the CRD housing to the RPV lower head and are part of the fuel vertical load support. Cracking of the weld connecting the RPV to the stub tube or the CRD housing to the stub tube could result in leakage of the reactor coolant.

Content Deleted -
EPRI Proprietary Information

5.3.5 RPV Attachment Bracket Cracking

After the first cycle, the Alloy 600 steam dryer support brackets of a domestic BWR were examined [58]. Remote underwater video viewing first revealed indications of cracking in the 184° bracket.

Content Deleted -
EPRI Proprietary Information

5.4 Example Evaluation Related to Past Inspection Findings

The example evaluation presented in this section is that conducted to provide justification for continued operation without repair of the Quad Cities Unit 2 (QC2) RPV head following the discovery of cracking described earlier. The evaluation was conducted by GE Nuclear Energy and is described in detail in Reference 63. Since the evaluation was performed prior to the completion of the metallurgical analysis, conservative assumptions were made regarding the failure mechanisms in the clad and in the underlying base metal. The evaluation consisted of three main steps:

Content Deleted -
EPRI Proprietary Information

5.4.1 Applied Stress

Stresses in the region of cracking are due to (1) clad stress, (2) bolt preload stress, (3) pressure and thermal stress and (4) weld residual stress near the vessel seam weld.

Content Deleted -
EPRI Proprietary Information

Content Deleted -
EPRI Proprietary Information

Content Deleted -
EPRI Proprietary Information

5.4.2 Environmental Crack Growth

Stress corrosion cracking of low alloy steels can occur under constant load or monotonically increasing load in a high temperature environment. In addition to instances of SCC in the laboratory, there have been several field cracking incidences of SCC in other materials with extension into the low alloy steel material (see Section 5.3). In the field observations, crack initiation has occurred in other susceptible materials such as stainless steel clad or Alloy 182 welds. Residual stresses due to the clad or welds or localized PWHT have also contributed to crack growth. For the purpose of this analysis, a conservative assumption of SCC crack growth in the low alloy steel is used. BWRVIP-60 [40] also presents a methodology for evaluating crack growth.

The SCC propagation rate in low alloy steels is a complex function of:

Content Deleted -
EPRI Proprietary Information

Content Deleted -
EPRI Proprietary Information

The crack growth analysis described here provides a reasonable prediction for the crack depth (as confirmed by comparison with the actual measured depth). Therefore, the predictive model can be used for determining the depth at the end of the next fuel cycle in the Section XI IWB-3600 analysis described in the next section.

5.4.3 Section XI Fracture Margin Assessment

Content Deleted -
EPRI Proprietary Information

Content Deleted -
EPRI Proprietary Information

Content Deleted -
EPRI Proprietary Information

Content Deleted -
EPRI Proprietary Information

Content Deleted -
EPRI Proprietary Information

5.5 Example Acceptance by Repair

An example of acceptance by repair of service induced degradation of a BWR RPV nozzle is presented in this Section. An Alloy 82 weld overlay repair has been developed for application to a low alloy steel nozzle to stainless steel or Alloy 600 safe end joint subject to inservice degradation from IGSCC [71]. The repair approach consists of a full structural weld overlay, using automatic gas tungsten arc welding technique deposited in accordance with a temperbead welding approach similar to that presented in ASME Section XI Code Case N-432.

Content Deleted -
EPRI Proprietary Information

**Content Deleted -
EPRI Proprietary Information**

5.6 10CFR50 Appendix G Equivalent Margin Analysis for Low Upper Shelf Energy (USE) in BWR/2 through BWR/6 Vessels

10CFR50 Appendix G requires that 50 ft-lb upper shelf energy (USE) be maintained in the vessel beltline low alloy steel material throughout operation. It further requires, if 50 ft-lb USE cannot be demonstrated, that methods to show equivalent margin be provided. As a result of responses to Generic Letter 92-01 [72], the Owners' Group performed an equivalent margin analysis, following the methods provided in the Code Case N-512. The equivalent margin analysis was performed for all BWR/2 through BWR/6 vessels. The scope of the BWROG equivalent margin analysis has been to bound the materials and 32 effective full power year (EFPY) fluences of all U.S. BWR/2 through BWR/6 (BWR/2-6) vessels.

**Content Deleted -
EPRI Proprietary Information**

5.7 RPV Surveillance Program Requirements

Part of the effort to assure reactor vessel integrity involves evaluation of the fracture toughness of the vessel ferritic materials. The key values which characterize a material's fracture toughness are the reference temperature of nil-ductility transition (RT_{NDT}) and the upper shelf energy (USE). These are defined in 10CFR50 Appendix G [76] and in Appendix G of the ASME Boiler and Pressure Vessel Code, Section XI [77].

Appendix H of 10CFR50 [78] and ASTM E185 [79] establish the schedule and methods to be used for surveillance of the reactor vessel materials. There are typically three capsules installed in the reactor; the first two are typically removed and tested during the 32 effective full-power year (EFPY) design life. The capsule withdrawal schedule should permit monitoring of long- term effects. Although the withdrawal schedule is in terms of EFPY of the vessel with a design life of 32 EFPY, there is sufficient flexibility with BWRs that one of the three capsules could be used for license renewal. The first capsule is scheduled for withdrawal early in the vessel life to verify the initial predictions of the surveillance material response to the actual radiation environment. It is removed when the predicted shift exceeds the expected scatter by sufficient margin to be measurable. The exposure of the second capsule should not exceed the peak end-of-life (EOL) fluence on the inside surface of the vessel and the final capsule should not exceed twice the EOL vessel inside surface peak fluence. Since the fluence for BWR vessels are typically low, pulling the first capsule too early would result in a predicted shift that is not distinguishable from the data scatter. The second capsule would need to be pulled near EOL to achieve fluences similar to the EOL vessel inside surface peak fluence, leaving the third capsule available for license renewal.

The surveillance capsules contain flux wires for neutron flux monitoring and Charpy V-Notch impact and tensile test specimens. The Charpy V-Notch impact and tensile test specimens are fabricated using base metal from the beltline region, as well as weld metal from a similar heat of material as the beltline welds. The Charpy V-Notch impact and tensile specimens are to be tested to establish properties for the irradiated materials.

The results of the surveillance specimen testing are presented in a report, as required per 10CFR50 Appendices G and H [76, 78]. The irradiated material properties are compared to available unirradiated properties to determine the effect of irradiation on material toughness for the base and weld materials, through Charpy testing. Irradiated tensile testing results are provided and are compared with unirradiated data to determine the effect of irradiation on the stress-strain relationship of the materials.

Pressure-temperature (P-T) curves are included in the report to developed the steam dome pressure versus minimum vessel metal temperature incorporating appropriate non-beltline limits and irradiation embrittlement effects in the beltline. The P-T curves are established to the requirements of 10CFR50, Appendix G [76] to assure that brittle fracture of the reactor vessel is prevented. Recent ASME Code cases have made provisions for reducing conservatism in the P-T curves. The elimination of postulating an axial flaw in circumferential welds will reduce conservatism for plants that have limiting circumferential welds. In addition, the introduction of using K_{IC} rather than K_{IA} will result in a general reduction of the required temperature for all P-T curves.

Recently, the utilities of the BWRVIP have decided to develop an integrated surveillance program (ISP). Implementing a BWR ISP will not only improve the quality of data and compliance of the entire BWR fleet, but will also provide data that better represent the BWR materials and operating conditions.

The proposed integrated program will review and assess all available surveillance capsule data – both the "existing program" capsules currently in each reactor and the additional capsules that were developed for the Supplemental Surveillance Program (SSP). The proposed integrated program will be based on those capsules which best meet the needs of the BWR fleet. Capsule data that provide little or no added value will not be included and need not be tested because other materials in the integrated program (including the SSP) will be used to replace these capsules.

Table 5-1
Critical Through-Wall Flaw Sizes in Beltline Region

Content Deleted -
EPRI Proprietary Information

Content Deleted -
EPRI Proprietary Information

Figure 5-1
Crack Profile - Chinshan 2 N2E Recirculation Inlet Nozzle [62]

Content Deleted -
EPRI Proprietary Information

Figure 5-2
Normal Operation Stress Distribution in Cracked Top Head Region

Content Deleted -
EPRI Proprietary Information

Figure 5-3
Theoretical and Observed Crack Propagation Rate/Stress Intensity Relationships for 0.010% Steel in Oxygenated Water for Corrosion Potentials Between 0 and -100mV SHE

Content Deleted -
EPRI Proprietary Information

Figure 5-4
Comparison of Models for Computing K versus a

Content Deleted -
EPRI Proprietary Information

Figure 5-5
Top Head Crack Growth for ½ Inch Clad, 15 ksi Clad Stress

Content Deleted - EPRI Proprietary Information

Figure 5-6
K versus a for Pressure Test Condition, Cracked Top head Region (the limiting case)

Content Deleted - EPRI Proprietary Information

Figure 5-7
Top Head SCC Growth Predicted for Next Cycle of Operation

Content Deleted -
EPRI Proprietary Information

Figure 5-8
Primary Stress Distribution for Grindout Geometry

Content Deleted -
EPRI Proprietary Information

Figure 5-9
K versus a for Pressure Test Condition, Beltline Region (Axial Flaw)

Content Deleted -
EPRI Proprietary Information

Figure 5-10
Brunswick 1 Nozzle to Safe-End Weld Overlay Schematic [61]

6
REFERENCES

[1] American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code- "Boiler and Pressure Vessel Code of Design and Construction Practices: Section XI," ASME, 345 East 47th Street, New York, NY 10017, 1989 Edition

[2] "BWR Vessel and Internals Project, Safety Assessment of BWR Reactor Internals (BWRVIP-06)," EPRI Report TR-105707, October 1995.

[3] "BWR Vessel and Internals Project, BWR Core Shroud Inspection and Flaw Evaluation Guidelines, Revision 2 (BWRVIP-01)," EPRI Report TR-107079, October 1996.

[4] "BWR Vessel and Internals Project, BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines (BWRVIP-18)," EPRI Report TR-106740, July 1996.

[5] "BWR Vessel and Internals Project, BWR Core Plate Inspection and Flaw Evaluation Guideline (BWRVIP-25)," EPRI Report TR-107284, December 1996.

[6] "BWR Vessel and Internals Project, BWR Top Guide Inspection and Flaw Evaluation Guidelines (BWRVIP-26)," EPRI Report TR-107285, December 1996.

[7] "BWR Vessel and Internals Project, BWR Standby Liquid Control System/Core Plate ΔP Inspection and Flaw Evaluation Guidelines (BWRVIP-27)," EPRI Report TR-107286, April 1997.

[8] "BWR Vessel and Internals Project, BWR Shroud Support Inspection and Flaw Evaluation Guidelines (BWRVIP-38)," EPRI Report TR-108823, September 1997.

[9] "BWR Vessel and Internals Project, BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines (BWRVIP-41)," EPRI Report TR-108728, October 1997.

[10] "BWR Vessel and Internals Project, BWR Lower Plenum Inspection and Flaw Evaluation Guidelines (BWRVIP-47)," EPRI Report TR-108727, December 1997.

[11] "BWR Vessel and Internals Project, Vessel ID Attachment Weld Inspection and Evaluation Guidelines (BWRVIP-48)," EPRI Report TR-108724, February 1998.

[12] "BWR Vessel and Internals Project, Instrument Penetration Inspection and Flaw Evaluation Guidelines (BWRVIP-49)," EPRI Report TR-108695, March 1998.

[13] "BWR Vessel and Internals Project, Shroud Vertical Weld Inspection and Evaluation Guidelines(BWRVIP-63)," EPRI Report TR-113170, June 1999.

[14] "BWR Feedwater Nozzle and Control rod Drive Return Line Nozzle Cracking," NUREG-0619, U.S. NRC, November 1980.

[15] "Technical Report on Materials Selection and Processing Guidelines for BWR Coolant Boundary Piping - Final Report," NUREG-0313, Rev. 2, U.S. Nuclear Regulatory Commission, January 1988.

[16] "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," EPRI Report TR-105697, September 1995.

[17] "BWR Vessel Shell Welds Inspection Recommendations (BWRVIP-05)," NRC Staff Presentation to the ACRS on July 9, 1998.

[18] Pinelli, RA, "Distribution of BWROG Model Inspection Program Draft Document," BWROG-94099, July 25, 1994.

[19] Beck, GJ, "Recommendations on Inservice Inspection of BWR Internal Components, Attachments and Penetrations," BWROG-91028, April 4, 1991.

[20] "BWR Feedwater Nozzle/Sparger Final Report, Supplement 2," GENE NEDE-21821-02, August 1979.

[21] EPRI Report NP-4273-SR, "Boiling Water Reactor Owners Group Intergranular Stress Corrosion Cracking Research Program: Executive Summary, Phase 2 (1984-1988)," August 1990.

[22] "Status Report: Intergranular Stress Corrosion Cracking of BWR Core Shrouds and Other Internals," NUREG-1554, US NRC, March 1996.

[23] "BWR RPV License Renewal Industry Report, Revision 1," EPRI Report TR-103836, July 1994.

[24] "10CFR50 Appendix G Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2 -6 Vessels," GE Report NEDO-32205-A, Revision 1, February 1994.

[25] "BWR Vessel and Internals Project, Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity Issues (BWRVIP-08NP)," EPRI Report TR-105908NP, November 1995.

[26] "BWR Vessel and Internals Project, Update of Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity Issues (BWRVIP-46)," EPRI Report TR-109727, December 1997.

[27] "Environmental Fatigue Evaluations of Representative BWR Components," EPRI Report TR-107943, May 1998.

[28] "Evaluation of Environmental Thermal Fatigue Effects on Selected Components in a BWR Plant," EPRI Report TR-110356, April 1998.

[29] "Reactor Pressure Vessel Attachment Welds: Degradation Assessment," EPRI NP-7139-D, Project T303-3 Final Report, May 1991.

[30] "Reactor Pressure Vessel Internals Study", EPRI Repair Applications Center, June 1989.

[31] "Evaluation of Safe-end Weld Materials and Safe-end Replacement Experience," EPRI NP-4443, Project T303-3 Interim Report, February 1986.

[32] SECY-94-245.

[33] "Boiling Water Reactor Feedwater Nozzle/Sparger Final Report," GE Report No. NEDE-21821-A, Class III (February 1980).

[34] GE Proprietary Information: GENE Power Uprate Analyses.

[35] NRC Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel", May 1973.

[36] Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," January 25, 1988.

[37] "BWR Water Chemistry Guidelines, 1996 Revision," BWRVIP-29, EPRI Report TR-103515-R1, December 1996.

[38] "BWR Vessel and Internals Project, Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection (BWRVIP-62)," EPRI Report TR-108705, December 1998.

[39] P.L. Andresen, "Irradiation Assisted Stress Corrosion Cracking", in Book on Stress Corrosion Cracking: Materials Performance and Evaluation, Ed. R.H. Jones, ASM, Materials Park, 1992, p. 181-210.

[40] "BWR Vessel and Internals Project, Evaluation of Stress Corrosion Crack Growth in Low Alloy Steel Vessel Materials in the BWR Environment (BWRVIP-60)," EPRI Report TR-108709, March 1999.

[41] F. P. Ford, "Environmentally Assisted Cracking in Low Alloy Steels," EPRI NP-7473L, January 1992.

[42] NRC Final SER - of the BWR Vessel and Internals Project BWRVIP-60 Report (TAC No. MA5057) dated July 8, 1999.

[43] "Generic License Renewal Document Issues Summary", EPRI Report TR-107521, April 1998.

[44] US NRC Regulatory Guide 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations", Revision 1 dated February 1983.

[45] US NRC IE Bulletin No. 82-03, "Stress Corrosion Cracking in Thick-Wall, Large-Diameter, Stainless Steel, Recirculation System Piping At BWR Plants", dated October 14, 1982 and Revision 1 dated October 28, 1982.

[46] US NRC IE Bulletin No. 83-02, Stress Corrosion Cracking in Large-Diameter Stainless Steel Recirculation System Piping At BWR Plants", dated March 4, 1983.

[47] "Alternative BWR Feedwater Nozzle Inspection Requirements," GE Report No. GE-NE-523-A71-0594, Rev. 0 (October 1995), prepared by BWROG.

[48] Letter from T. Essig to T.J. Rausch, dated June 5, 1998, Subject: "BWROG - Safety Evaluation of Proposed Alternative to BWR Feedwater Nozzle Inspections," (TAC No. 94090).

[49] BWROG Reference on ISI Training Personnel.

[50] NRC Final SER - of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC No. M93925) dated July 28, 1998.

[51] NRC Generic Letter 94-03, dated July 25, 1994, "Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors."

[52] "BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines (BWRVIP-03) Revision 1" EPRI Report TR-105696-R1, March 1999.

[53] "BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Nickel Base Austenitic Alloys in RPV Internals (BWRVIP-59)," EPRI Report TR-108710, December 1998.

[54] ASME Boiler & Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components", Paragraph IWB-3600 and Appendix A, Analysis of Flaw Indications.

[55] T.U. Marston, et al., "Flaw Evaluation Procedures: ASME Section XI," Electric Power Research Institute Report, EPRI NP-710-SR, August 1978.

[56] W.H. Bamford, "Technical Basis for Revised Reference Crack Growth Rate Curves for Pressure Boundary Steels in LWR Environment," Journal of Pressure Vessel Technology, Trans. of ASME, Vol. 102, p. 433, November 1980.

[57] "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," US Nuclear Regulatory Commission Report, NUREG-0619, November 1980.

[58] B.M. Gordon, D.E. Delwiche and G.M. Gordon, "Service Experience of BWR Pressure Vessels," ASME PVP Volume 119, June 1987.

[59] C.J. Czajkowski, "Evaluation of Type 182 Weld Metal Cracks in the Pilgrim Nuclear Power Station", Corrosion 86, Paper No. 253 (March 1986).

[60] R.E. Smith, R. Hanford and S.C. Cheng, "Pressure Vessel Nozzle Repair," Nuclear Engineering and Design 124 (1990), 79-89.

[61] "Metallurgical Evaluation of Safe-End-To Nozzle Ring Removed from Brunswick Steam Electric Plant (Unit 1) Recirculation Inlet Nozzle," GE Report No. NEDC-32020, Class II, October 1992.

[62] S.C. Cheng, "Overview of Chinshan Nuclear Power Station Recirculation Pipe Repair and Replacement," NUREG/CP-0109 (1990), pp. 299-336.

[63] "Evaluation of Reactor Pressure Vessel Head Cracking in Two Domestic BWRs," EPRI Report TR-101971, February 1993.

[64] NRC Generic Letter 81-11, dated February 20, 1981.

[65] Bamford, et.al., "Stress Intensity Factors for Continuous Surface Flaws in Reactor Pressure Vessels," ASTM STP 590, February 1976.

[66] Tada, et.al., "The Stress Analysis of Cracks Handbook," Paris Productions, St. Louis, 2^{nd} Edition, 1985.

[67] Paris and Shih, "Stress Analysis of Cracks," ASTM STP 381 (1965).

[68] "Stress Analysis of Closure," Babcock & Wilcox Company, June 1969, GE VPF #1744-211.

[69] "Pressure Temperature Curves per Regulatory Guide 1.99, Revision 2 for the Dresden and Quad Cities Nuclear Power Stations," GE Report No. SASR 89-54, DRF 137-0010, August 1989.

[70] A. Zahoor, "Closed Form Expressions for Fracture Mechanics Analysis of Cracked Pipes," Journal of Pressure Vessel Technology, Vol. 107, May 1985.

[71] "Inconel Weld-Overlay Repair for Low-Alloy Steel Nozzle to Safe-End Joint," EPRI Report No. NP-7085-D, January 1991.

[72] NRC Generic Letter 92-01, Revision 1, dated March 6, 1992.

[73] "10CFR50 Appendix G Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2 Through BWR/6 Vessels," GENE NEDO-32205-A, February 1994.

[74] Letter from J.T. Wiggins to L. A. England, dated December 8, 1993: NRC Acceptance of and Transmittal of SER on Topical Report NEDO-32205, Revision 1.

[75] "N-16 Nozzles Upper Shelf Energy Evaluation", Altran Technical Report No. 96124-TR-01, Rev. 0, prepared for Carolina Power and Light Company, December 1996.

[76] "Fracture Toughness Requirements," Appendix G to Part 50 of Title 10 of the Code of Federal Regulations, December 1995.

[77] "Protection Against Non-Ductile Failure," Appendix G to Section XI of the 1989 ASME Boiler & Pressure Vessel Code.

[78] "Reactor Vessel Material Surveillance Program Requirements," Appendix H to Part 50 of Title 10 of the Code of Federal Regulations, December 1995.

[79] "Surveillance Test for Nuclear Reactor Vessels," Annual Book of ASTM Standards, E185, 1982 edition.

A
BWR REACTOR PRESSURE VESSEL COMPONENTS DEMONSTRATION OF COMPLIANCE WITH TECHNICAL INFORMATION REQUIREMENTS OF THE LICENSE RENEWAL RULE (10 CFR 54.21)

The purpose of Appendix A is to demonstrate that this report provides the necessary information to comply with the technical information requirements pursuant to Paragraphs 54.21 [a] and [c], and 54.22, and the NRC's findings under 54.29[a] of the license renewal rule [A1]. It is intended that the NRC's review and approval of Appendix A will allow utilities the option to incorporate the report and Appendix by reference in a plant-specific integrated plant assessment (IPA) and time-limited aging analysis (TLAA) evaluation. If a license renewal applicant confirms that this report applies to their plant's current licensing basis (CLB) and that the results of the Appendix A IPA and TLAA evaluation are in effect at their plant, then no further review by the NRC of the matters described herein is needed. The Appendix structure, which was used in the Vessel I&E Guideline, is based on discussions of degradation mechanisms instead of the effects of a particular degradation mechanism.

A.1 Description of the BWR Reactor Pressure Vessel Components and Intended Functions

The reactor pressure vessel (RPV) components of the BWR consist of a vertical cylindrical vessel of welded low alloy steel construction with a removable top head (flanged and joined via the head to flange closure studs), closure studs, a bottom head which is welded to the vertical shell, multiple nozzles and safe-ends, multiple penetrations and control rod drive stub tubes, a vessel support skirt and numerous attachment welds. The specifics of the vessel as they apply to the BWRVIP plants and the vessel components are given in the main report.

A.1.1 Pressure Vessel Structure and Supports

The function of the vessel shell, top head, bottom head and flange with the closure studs is to (1) form a pressure boundary to contain reactor coolant /moderator and against leakage of radioactive materials into the drywell, and (2) provide structural support for the reactor core and internals. BWR vessels have been manufactured using different low alloy steel materials.

Content Deleted -
EPRI Proprietary Information

**Content Deleted -
EPRI Proprietary Information**

A.1.2 Vessel Nozzles and Safe-ends

The intended function of the nozzles and safe-ends is to form a pressure boundary to contain reactor coolant /moderator and against leakage of radioactive materials into the drywell. Reactor vessels have many nozzle penetrations for piping and equipment.

**Content Deleted -
EPRI Proprietary Information**

The safe-ends are attached to each of the nozzles, providing a transition to the piping material. The material of the safe-end varies from system to system as well as from plant to plant, which is a result of the RPV vendor's typical practice or replacement efforts to remove IGSCC susceptible materials and/or configurations.

**Content Deleted -
EPRI Proprietary Information**

A.1.3 Vessel Internal Attachments

The intended function of the attachment welds is to provide structural support for the reactor core and internals. The RPV attachments include the brackets used to hold internal components such as the core spray piping and the jet pump riser piping as well as the shroud support structure. The most significant attachments are those joining the shroud support structure to the RPV pressure boundary. The welds are generally made using Alloy 82/182 material.

A.1.4 Instrument, In-core Housing, CRD Stub Tube and CRD Housing Penetrations

There are several different penetrations associated with the RPV. The intended function of the penetrations is to form the pressure boundary to contain the reactor coolant/moderator and against leakage of radioactive materials into the drywell. The CRD stub tubes, in addition, function to provide structural support for the reactor core and internals. The specific penetrations include the instrument nozzles required to measure water level, the core plate ΔP/standby liquid control (SLC) system. The in-core monitor housings (ICMH), CRD stub tubes and CRD housings.

**Content Deleted -
EPRI Proprietary Information**

A.2 Reactor Pressure Vessel Components Subject to Aging Management Review

Paragraph 54.21(a)(1.) of the rule provides the requirements for identifying the RPV components that are subject to aging management review. To satisfy the requirements of 54.21(a)(1.), the guidance provided in the NEI industry guideline [A2] was used to identify passive components and then to identify those that are long-lived. For the vessel components, a screening methodology was not needed to make this determination. All of the components considered to make up the vessel or that are attached to the vessel are passive and long-lived and are subject to aging management review.

A.3 Management of Aging Effects (54.21 [a] [3])

A.3.1 Description of Aging Effects

For the purpose of this Appendix, the BWR Reactor Pressure Vessel Industry Report [A3] and the resolution to the NRC's questions on the Industry Report are used to identify the aging mechanisms for the vessel components. Aging mechanisms are the causes of the aging effects. NUREG 1557 [A4] is used to establish the correlation between the aging effects and their associated aging mechanisms. If the industry report and this BWRVIP document conclude that the aging mechanism is significant, then the associated aging effect is included in this aging management review. Using this methodology, it was determined that neutron embrittlement, fatigue crack initiation and growth, and stress corrosion crack initiation and growth are the aging effects that require aging management review for the vessel components. This conclusion is consistent with the scope and intent of this report.

The causes of neutron embrittlement are provided in Section 3.1. The causes and locations of importance for fatigue crack initiation and growth are given in Section 3.2. Finally, the causes of stress corrosion crack initiation and growth are provided in Sections 3.3, 3.4 and 3.5. The susceptibility factors of environment, materials, and stress state are also discussed in these sections. A discussion of the potential locations degraded by this mechanism is presented for each of the vessel components in Subsections 3.1 through 3.3. The applicant will need to demonstrate that: (a) the safety assessment of individual vessel components, and (b) the evaluation of the safety consequences of individual vessel component failure, applies to their plant's CLB.

A.3.2 Assessment of Aging Effects and Programs

The vessel component performance history is described in Section 5 of this report, as well as in other BWRVIP I&E documents. Using this information, four specific criteria were used in developing the inspection strategy: (1) safety significance; (2) detectability of failure or cracking; (3) field cracking history; and (4) prior inspections. The inspection criteria are primarily governed by safety considerations.

The review of the field cracking indicates that the principal locations of cracking have been thermal fatigue cracking in the blend radius of feedwater nozzles and IGSCC in the nozzle to safe end weld butters made with Alloy 182. Corrective actions have been taken to reduce thermal stresses and to replace or repair susceptible materials. The rest of the vessel components have not experienced significant field cracking. Numerous ongoing mandated inspections (Table 4-1) provide confidence that the field experience gives an accurate assessment of the materials' behavior. There are also mandated surveillance programs to assess and manage reduction in beltline toughness.

Content Deleted -
EPRI Proprietary Information

BWR owners are required to schedule the inspections to meet the specified time frames. Inspection may be conducted in combination with other maintenance activities (e.g., control rod replacement or relocation, fuel redistribution or replacement).

During inspection, if one or more flaws are found, the licensee shall make the appropriate decision for follow-on inspections, scope expansion, or other actions in accordance with the ASME Code, NUREG documents and BWRVIP documents.

Content Deleted -
EPRI Proprietary Information

A.3.3 Demonstration that the Effects of Aging are Adequately Managed

The effects of neutron embrittlement leading to reduced material toughness, cyclic loading leading to fatigue and corrosion and stress corrosion cracking are the degradation mechanisms for the vessel components that require aging management review for license renewal. These aging effects

Content Deleted - EPRI Proprietary Information

A.4 Time Limited Aging Analyses (54.21 [c] [1])

The six criteria contained in the NEI industry guideline [A2] were applied to identify the time limited aging analysis (TLAA) issues; that is, those calculations and analyses that:

Involve the reactor pressure vessel components.

Consider the effects of aging.

Involve time-limited assumptions defined by the current operating term.

Were determined to be relevant in making a safety determination.

Involve conclusions or provide the basis for conclusions related to the capability of the vessel components to perform their intended function.

Are incorporated or contained by reference in the CLB.

Based on these criteria, several generic categories of TLAAs need to be addressed. These categories and the process for preparing the TLAA are given in the following sections.

Content Deleted - EPRI Proprietary Information

A.4.1 Pressurization Temperature (P-T) Curves

Nuclear Regulatory Commission (NRC) 10CFR50 Appendix G [A5] specifies fracture toughness requirements to provide adequate margins of safety during the operating conditions that a pressure-retaining component may be subjected over its service lifetime. The ASME Code (Appendix G of Section XI of the ASME Code [A6]) forms the basis for the requirements of 10CFR50 Appendix G. The limits for pressure and temperature are required by 10CFR50 Appendix G for three categories of operation: (1) hydrostatic pressure tests and leak tests, (2) core not critical heatup/cooldown, and (3) core critical operation. The condition that results in the highest temperature for the limiting material determines the minimum temperature requirement for the vessel.

Content Deleted -
EPRI Proprietary Information

A.4.2 Equivalent Margin Analysis

One key effect of irradiation on the reactor pressure vessel plate and weld materials is defined and measured by Upper Shelf Energy (USE). An evaluation was performed [A7] to provide a 40-year or 32 effective full power years (EFPY) generic equivalent margin analysis to be used for any plant unable to meet 10CFR50 Appendix G requirements for USE for a plant life. Due to plant license renewal, this evaluation has been performed for a plant life of 60 years or 54 EFPY.

Content Deleted -
EPRI Proprietary Information

A.4.3 Generic TLAAs with Cumulative Usage Factors Less Than 0.5

Fatigue is the behavior of a material under the action of repeated fluctuating stresses or strains. After a sufficient number of fluctuations, the stresses and strains can result in the initiation of the crack (the onset of which is defined as fatigue life). Fatigue usage for the components of the reactor pressure vessel (RPV) were evaluated consistent with the ASME Code Section III, NB-3222.4. Each component in the RPV is evaluated for fatigue and documented in the certified stress report. If the component was modified, then a modified stress report supersedes the certified stress report for the portion of the component that was modified.

The process for evaluating the license renewal fatigue usage is shown on Figures A-2 and A-3. As shown in Figure A-2, the first step (Step 1) for the fatigue evaluation process is to make a list of the components that were evaluated in the original certified and modification stress reports. A representative list of those components was developed from fatigue usage factors from eleven plants (Table A-3). Some variation in the fatigue usage value is due to the differences in the design of the nozzles and the details used for the evaluation. Note that the original intent of certification was to demonstrate that the fatigue usage was less than 1.0, so evaluations with a simple combination of the thermal cycles into a single limiting event were performed. Therefore, the relative differences between the fatigue usage values may not accurately represent the actual relative fatigue usage for the component.

The fatigue usage for several RPV components were originally evaluated using the ASME Code criterion for evaluating vessels "Not Requiring Analysis for Cyclic Operation" (i.e., "Exempt from Fatigue"). These components require no further evaluation, since even with an increase in life from 40 years to 60 years, the fatigue usage on these components will be insignificant.

Content Deleted -
EPRI Proprietary Information

A.4.4 Process for TLAAs with High Cumulative Usage Factors (> 0.5)

To calculate the License Renewal Fatigue Cumulative Usage (LRCUF), the original fatigue usage (OCUF) is simply multiplied by the ratio of the license renewal life to the current life (i.e., 60 years/40 years = 1.5). If the License Renewal Fatigue Usage is less than 1.0, then no further evaluation of the component is required.

If the fatigue usage exceeds 1.0, then there are four alternatives for further disposition of those components [see Figure A-3 (Step 5)]. Any one of the four alternatives is an acceptable method to disposition the component and, if one of the three calculation methods does not result in a fatigue usage less than 1.0, the other two alternatives may. If no alternative results in a fatigue usage less than 1.0, then Step 5d can be used to disposition the component.

Content Deleted -
EPRI Proprietary Information

Content Deleted -
EPRI Proprietary Information

A.4.5 Circumferential Weld Inspection Relief

Content Deleted -
EPRI Proprietary Information

Content Deleted - EPRI Proprietary Information

A.5 Exemptions (54.21 [c] [21])

Exemptions associated with vessel components that contain TLAA analysis issues will be identified and evaluated for license renewal by individual applicants.

Content Deleted - EPRI Proprietary Information

A.6 Technical Specification Changes or Additions (54.22)

There will be changes to the technical specifications associated with the vessel as a result of this aging management review to ensure that the effects of neutron embrittlement are adequately managed. These changes will be made at the appropriate time prior to the end of the current license term. No other changes or additions are required.

A.7 Demonstration that Activities Will Continue to Be Conducted in Accordance with the CLB (54.29[a])

Sections A.1, A.2, and A.3 address the requirements 54.21 (a) of the rule. The vessel components that are subject to aging management review are identified, and it is demonstrated that the effects of aging are adequately managed. Sections A.4 and A.5 address the requirements of 54.21(c) of the rule. Plant-specific time limited aging analyses (TLAAs) and exemptions that require evaluation will be evaluated by the applicant. Section A.6 addresses the requirements of 54.22 of the rule.

Content Deleted - EPRI Proprietary Information

A.8 References

(Al) Title 10 of the Code of Federal Regulations, Part 54, "Requirements for License Renewal of Operating Licenses for Nuclear Power Plants," (60 Federal Register 2246 1), May 8, 1995.

(A2) Nuclear Energy Institute Report NEI 95-10 (Rev. 0), Industry Guideline for Implementing the Requirements of 10 CFR Part 54 the License Renewal Rule.

(A3) "BWR RPV License Renewal Industry Report, Revision 1," EPRI Report TR-103836, July 1994.

(A4) NUREG-1557, Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal, October 1996.

(A5) "Fracture Toughness Requirements," Appendix G to Part 50 of Title 10 of the Code of Federal Regulations, December 1995.

(A6) "Protection Against Non-Ductile Failure," Appendix G to Section XI of the 1989 ASME Boiler & Pressure Vessel Code.

(A7) H.S. Mehta, T.A. Caine, and S.E. Plaxton, "10CFR50 Appendix G Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2 Through BWR/6 Vessels," GENE NEDO-32205-A, Revision 1, February 1994.

(A8) Letter from J.T. Wiggins (NRC) to L.A. England (Gulf States Utilities Co.), "Acceptance for Referencing of Topical Report NEDO-32205, Revision 1, '10CFR50 Appendix G Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2 Through BWR/6 Vessels'", December 8, 1993.

(A9) D. Lurie and R.H. Moore, "Applying Statistics," NUREG-1475, February 1994.

(A10) Final SER of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC No. M93925), July 28, 1998.

(A11) NRC Generic Letter 98-05: Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Shell Welds, November 10, 1998.

(A12) Letter dated January 13, 1998, from Vaughn Wagoner/W. Bilanin to BWRVIP Committee Members, BWRVIP Response to NRC RAI on BWRVIP-05, October 10, 1997.

Table A-1
Summary of the 10CFR50 Appendix G Requirements

Operating Condition and Pressure	**Minimum Temperature Requirement**
I. Hydrostatic Pressure Test & Leak Test (Core is Not Critical) - Curve A	
1. At ≤ 20% of preservice hydrotest pressure	Larger of ASME Limits or of highest closure flange region initial RT_{NDT} + 60°F*
2. At > 20% of preservice hydrotest pressure	Larger of ASME Limits or of highest closure flange region initial RT_{NDT} + 90°F
II. Normal operation (heatup and cooldown), including anticipated operational occurrences	
A. Core not critical - Curve B	
1. At ≤ 20% of preservice hydrotest pressure	Larger of ASME Limits or of highest closure flange region initial RT_{NDT} + 60°F*
2. At > 20% of preservice hydrotest pressure	Larger of ASME Limits or of highest closure flange region initial RT_{NDT} + 120°F
B. Core critical - Curve C	
1. At ≤ 20% of preservice hydrotest pressure, with the water level within the normal range for power operation	Larger of ASME Limits + 40°F or of A.1 (Appendix G)
2. At > 20% of preservice hydrotest pressure	Larger of ASME Limits + 40°F or of A.2 (Appendix G) + 40°F or the minimum permissible temperature for the inservice system hydrostatic pressure test

* 60°F adder is included by GE as an additional conservatism, as discussed in Section 5.

Table A-2
Calculated 54 EFPY USE Values as a Function of RPV Plate and Weld Material

Content Deleted -
EPRI Proprietary Information

Table A-3
Representative Fatigue Usage Factors

**Content Deleted -
EPRI Proprietary Information**

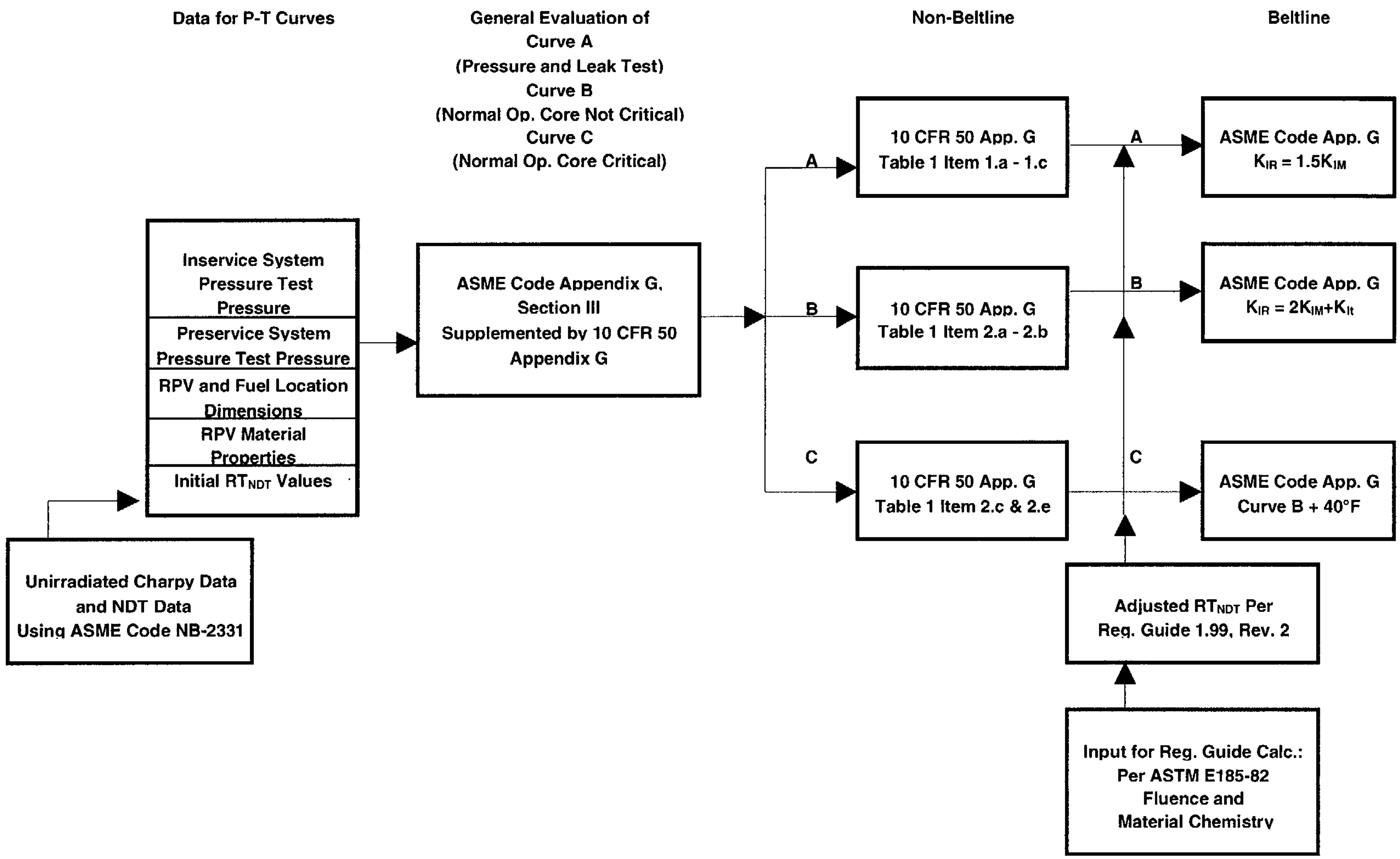

Figure A-1
Process to Develop P-T Curves in Compliance with 10CFR50 Appendix G

Content Deleted -
EPRI Proprietary Information

Figure A-2
TLAA - CUF Evaluation Process

Content Deleted -
EPRI Proprietary Information

Figure A-3
TLAA - CUF Evaluation Process for Step 5

B
BOUNDING USE ANALYSIS FOR PLANT LICENSING RENEWAL (UP TO 60 YEARS/54 EFPY)

The equivalent margin analysis presented in this report establishes the minimum USE limits for BWR/2-6 vessel beltline materials. The purpose of this Appendix is to demonstrate, for all U.S. BWR/2-6 vessels, that the USE values predicted for the BWR vessel materials will, for 54 EFPY of operation, remain higher than the allowable USE limits, as required in 10CFR50, Appendix G. This Appendix supplements Section 8 of Reference B-1, which provides the basis for the evaluation for 32 EFPY. For the reader's convenience, the basis is again described herein. In addition, this Appendix contains the Plant Applicability Verification Form for Equivalent Margin Analysis to be used for 54 EFPY.

Content Deleted -
EPRI Proprietary Information

B.1 BWR/2 Plate Materials

Content Deleted -
EPRI Proprietary Information

Content Deleted -
EPRI Proprietary Information

B.2 BWR/3-6 Plate Materials

Content Deleted -
EPRI Proprietary Information

Content Deleted -
EPRI Proprietary Information

**Content Deleted -
EPRI Proprietary Information**

B.3 BWR/2-6 Weld Materials

**Content Deleted -
EPRI Proprietary Information**

B.3.1 SMAW

**Content Deleted -
EPRI Proprietary Information**

Content Deleted -
EPRI Proprietary Information

B.3.2 ESW

Content Deleted -
EPRI Proprietary Information

Content Deleted -
EPRI Proprietary Information

B.3.3 Non-Linde 80 SAW

Content Deleted -
EPRI Proprietary Information

Content Deleted -
EPRI Proprietary Information

B.3.4 Linde 80 SAW

Content Deleted -
EPRI Proprietary Information

Content Deleted -
EPRI Proprietary Information

B.4 Summary of Evaluation

Content Deleted -
EPRI Proprietary Information

Content Deleted -
EPRI Proprietary Information

B.5 Plant-Specific Applicability

Content Deleted -
EPRI Proprietary Information

Content Deleted -
EPRI Proprietary Information

B.6 References

[B-1] H.S. Mehta, T.A. Caine, and S.E. Plaxton, "10CFR50 Appendix G Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2 Through BWR/6 Vessels," NEDO-32205-A, Revision 1, February 1994.

[B-2] D.J. Ayres, and R.E. Smith, "Statistical Analysis of Charpy-V Impact Properties SA533 Grade B Class 1 and SA516 Grade 70 Plate Material," Transactions of the ASME, February 1973.

[B-3] D. Lurie, and R.H. Moore, "Applying Statistics," U.S. Nuclear Regulatory Commission, February 1994.

[B-4] E.A. Eason, J.E. Wright, and E.E. Nelson, "Multivariable Modeling of Pressure Vessel and Piping J-R Data," NUREG/CR-5729, May 1991.

Table B-1
Data on BWR Beltline Materials for R.G. 1.99 USE Evaluation for 54 EFPY

Content Deleted -
EPRI Proprietary Information

Table B-2
Data on BWR Beltline Plates

Content Deleted -
EPRI Proprietary Information

Table B-3
Equivalent Margin Analysis Plant Applicability Verification Form
for ______________________________

Content Deleted -
EPRI Proprietary Information

Table B-4
Equivalent Margin Analysis Plant Applicability Verification Form
for ______________________

Content Deleted -
EPRI Proprietary Information

Table B-5
Equivalent Margin Analysis Plant Applicability Verification Form
for ______________________________

Content Deleted -
EPRI Proprietary Information

Content Deleted -
EPRI Proprietary Information

Figure B-1
BWR/2 Plates (Longitudinal) Meet Equivalent Margin Requirements

Content Deleted -
EPRI Proprietary Information

Figure B-2
BWR/2 Plates (Transverse) Meet Equivalent Margin Requirements

Content Deleted -
EPRI Proprietary Information

Figure B-3
BWR/3-6 Plates (Longitudinal) Meet Equivalent Margin Requirements

Content Deleted -
EPRI Proprietary Information

Figure B-4
BWR/3-6 Plates (Transverse) Meet Equivalent Margin Requirements

**Content Deleted -
EPRI Proprietary Information**

**Figure B-5
SMAW Materials Meet Equivalent Margin Requirements**

Content Deleted -
EPRI Proprietary Information

Figure B-6
ESW Materials Meet Equivalent Margin Requirements

Content Deleted -
EPRI Proprietary Information

Figure B-7
Non-Linde SAW Materials Meet Equivalent Margin Requirements

Target:
Nuclear Power

About EPRI

EPRI creates science and technology solutions for the global energy and energy services industry. U.S. electric utilities established the Electric Power Research Institute in 1973 as a nonprofit research consortium for the benefit of utility members, their customers, and society. Now known simply as EPRI, the company provides a wide range of innovative products and services to more than 1000 energy-related organizations in 40 countries. EPRI's multidisciplinary team of scientists and engineers draws on a worldwide network of technical and business expertise to help solve today's toughest energy and environmental problems.

EPRI. Powering Progress

TR-113596NP

EPRI • 3412 Hillview Avenue, Palo Alto, California 94304 • PO Box 10412, Palo Alto, California 94303 • USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com